Entergy

Entergy Nuclear Operations, Inc.
600 Rocky Hill Road
Plymouth, MA 02360

Pilgrim Nuclear Power Station

October 21, 2013

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D.C. 20555

SUBJECT: Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket No.: 50-293
License No.: DPR-35

Licensee Event Report 2013-008-00, Manual Scram - Reactor Feed Pump Trip

LETTER NUMBER: 2.13.082

Dear Sir or Madam:

The enclosed Licensee Event Report (LER) 2013-008-00, "Manual Scram - Reactor Feed Pump Trip" is submitted in accordance with 10 CFR 50.73.

This letter contains no commitments.

Please do not hesitate to contact me at (508) 830-8403, if there are any questions regarding this submittal.

Sincerely,

Joseph R. Lynch

Joseph R. Lynch
Licensing Manager

JRL/fm

Attachment 1: Licensee Event Report 2013-008-00, Manual Scram - Reactor Feed Pump Trip

IE22
NRR

cc:

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USNRC Senior Resident Inspector
Pilgrim Nuclear Power Station

Attachment 1
Letter Number 2.13.082

Licensee Event Report 2013-008-00

Manual Scram - Reactor Feed Pump Trip

(4 Pages)

NRC FORM 366 (10-2010) | U.S. NUCLEAR REGULATORY COMMISSION | APPROVED BY OMB: NO. 3150-0104 EXPIRES: 10/31/2013

LICENSEE EVENT REPORT (LER)

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4. TITLE
Manual Scram - Reactor Feed Pump Trip

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
									N/A	
08	22	2013	2013	008	00	10	21	2013	FACILITY NAME N/A	DOCKET NUMBER

9. OPERATING MODE: N

10. POWER LEVEL: 98%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☒ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	Specify in Abstract below or in NRC Form 366A

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
Joseph R. Lynch, Licensing Manager	(508)-830-8403

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
N/A									

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
☐ Yes (If yes, complete 15. *EXPECTED SUBMISSION DATE* ☒ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

On Thursday, August 22, 2013 at 0755 [EDT] with the reactor critical at 98% core thermal power (CTP), and the mode switch in RUN, the Pilgrim Nuclear Power Station (PNPS) was manually scrammed due to lowering reactor water level resulting from a trip of the reactor feed pumps. The reactor feed pumps tripped due to a loss of power to the pump seal cooling water flow switch relays and resultant automatic actuation of the feed pump trip circuit.

The direct cause of the reactor feed pump trip was an automatic actuation of the feed pump trip circuitry. The feed pump trip circuits actuated as designed in response to a loss of power to the pump seal cooling water flow switch relays. The seal cooling water flow switch relays lost power as a result of a 120V AC breaker trip resulting from a short-to-ground fault in the associated circuits fed by the breaker. Corrective action was taken to repair the ground fault in the associated circuits fed by the 120V AC breaker and to revise reactor feed pump trip circuit design to remove the loss of seal water trip function.

This event posed no threat to public health and safety.

NARRATIVE

BACKGROUND:

The condensate and feed water systems are described in Section 11.8 of the Updated Final Safety Analysis Report (UFSAR). The feed water system has a power generation objective to provide a dependable supply of feed water to the reactor during power operation. The system has no required safety functions and is not required to operate in the event of plant shutdown due to accidents and transients. The UFSAR does evaluate a plant transient event that involves the loss of all feed pumps during power operation.

The feed water system has three, one-third capacity, motor driven feed pumps. Each feed pump is provided with mechanical seals at each end of the pump casing. The seal design incorporates an internal pumping ring, which circulates water between the seal cavity and an external cooler. The external cooler relies on cooling water flow from the turbine building closed cooling water (TBCCW) system. The TBCCW cooling water to the external seal coolers passes through flow switches which, at the time of the event, provided both an alarm and feed pump trip function given a low flow of cooling water to the external coolers.

All TBCCW flow switch relays are powered from a single breaker from the Y1, 120V AC power supply panel.

In 2011, a modification was implemented to revise the configuration of the feed pump TBCCW flow switches and pump trip relays. The modification was initiated to address a scram trip reduction recommendation identified by the Boiling Water Reactor Owner's Group (BWROG) Scram Frequency Reduction Committee (SFRC) and was intended to reduce potential for plant scram. The plant specific modification implemented at Pilgrim Station revised the reactor feed pump trip relay logic design and inadvertently introduced a new configuration that initiates actuation of the feed pump trip logic on loss of power to the cooling water flow switch relays. Previously, a loss of power to the relays would not trip the feed pumps.

On August 22, 2013 during routine plant operation, a ground fault occurred on an electrical circuit powered from 120V AC power panel Y1, breaker #24. The breaker opened as designed in response to the ground fault. This resulted in a loss of power to the cooling water flow switch relays and subsequent reactor feed pump trip.

EVENT DESCRIPTION:

On Thursday, August 22, 2013 at 0755 [EDT] with the reactor critical at 98% core thermal power (CTP), and the mode switch in RUN, the Pilgrim Nuclear Power Station (PNPS) was manually scrammed due to lowering reactor water level resulting from a trip of the reactor feed pumps. The reactor feed pumps tripped due to a loss of power to the pump seal cooling water flow switches and automatic actuation of the feed pump trip circuit.

Based on lowering reactor water level, plant operators manually scrammed the reactor in accordance with plant operating procedures. Following the reactor scram, all rods were verified to be fully inserted. All 4 KV busses transferred to the startup transformer as designed. Reactor water level lowered to +12 inches initiating primary containment isolation Group II (reactor building isolation system (RBIS); standby gas treatment (SBGT)); and Group VI (reactor water cleanup (RWCU) system) systems automatically as per design. Reactor water continued to lower to (-)46 inches initiating primary containment system Group I isolation (main steam isolation valves (MSIVs); core standby cooling systems (CSCS) actuated which included automatic start and injection of the high pressure coolant injection (HPCI) system and the reactor core isolation cooling (RCIC) system; and an automatic start of the emergency diesel generators (EDGs) as per design. Reactor water level was promptly restored due to automatic actuation of the HPCI and RCIC

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systems.

CAUSE OF THE EVENT:

The direct cause of the feed pump trip event was a trip of 120V AC power panel Y1, Breaker #24 that resulted from a splice failure and short-to-ground in a conduit on a cable feeding solenoid valve, SV-3067, Moisture Separator Drain Tank T-103D spill valve. The Y1, Breaker #24 tripped per design and deenergized the reactor feed pump low seal water relays. The deenergized reactor feed pump low seal water relays actuated the feed pump trip logic. The root cause of the event is that single point vulnerability design criteria associated with the reactor feed pump low seal cooling water flow modification was not clearly defined and implemented.

CORRECTIVE ACTIONS:

Corrective action was completed to repair the identified short in the wiring feeding SV-3067 and implementation of a modification to remove the reactor feed pump low seal cooling water flow relay contacts from the reactor feed pump trip circuit.

Additional corrective actions are planned to:

- Review single point vulnerability modifications to ensure failure modes and effects analyses have been properly performed.

These and other actions are captured in the Corrective Action Program under Condition Report, CR-PNP-2013-5949.

ASSESSMENT OF SAFETY CONSEQUENCES:

The event posed no threat to public health and safety.

The event occurred during normal power operation while at 98% power with the mode switch in the "RUN" position. The reactor vessel pressure was approximately 1025 psig with reactor water temperature at saturation temperature for that pressure.

The core standby cooling systems (CSCS) consist of the high pressure coolant injection (HPCI) system, automatic depressurization system (ADS), core spray (CS) system, and the residual heat removal (RHR) system in the low pressure core coolant injection (LPCI) mode. Although not part of the CSCS, the reactor core isolation cooling (RCIC) system is capable of providing water to the reactor vessel for high pressure core cooling, similar to the HPCI System. These systems were operable at the time when the reactor feed pumps started to sequentially trip off line.

The reactor feed pumps provide make up flow to the reactor when the plant is operating. The loss of the feed pumps resulted in lowering reactor water level. The feed water system provides no safety function. The UFSAR addresses the initial core analysis involving a complete loss of feedwater.

A manual scram was initiated prior to automatic initiation of the analog trip system (ATS) scram logic that actuates on reactor low water level. After the manual scram, both the reactor water level low (+11.6") and low-low level (-46.3") limits were exceeded and ATS actuation logic initiated start up of the CSCS safety systems (HPCI and RCIC). Group II and VI isolation groups actuated, and reactor building isolation system (RBIS) and standby gas treatment system (SBGTS) actuated. Group I isolation logic actuated closing the main steam isolation valves (MSIVs) and other Group I valves. Recirculation pump drive motor breakers tripped and emergency diesel generators (EDGs) started. The EDGs did not load on the emergency buses which were powered from the preferred offsite power source (i.e., startup transformer). The plant safety system response was evaluated and determined to be consistent with design and in accordance with the event described in the UFSAR.

Given the loss of all feedwater, the conditional core damage probability (CCDP) was 4.69E-7 and the conditional large early release probability (CLERP) was 1.95E-7.

REPORTABILITY:

This report is submitted in accordance with 10 CFR 50.73(a)(2)(iv)(A) – Any event or condition that resulted in manual or automatic actuation of any system listed in paragraph 10 CFR 50.73 (a)(2)(iv)(B). The Reactor Protection System (RPS) including: reactor scram or trip is included in 10 CFR 50.73 (a)(2)(iv)(B).

This event was initially reported to the NRC in accordance with 10 CFR 50.72(b)(2)(iv)(A); 50.72(b)(2)(iv)(B); and 50.72(b)(3)(iv)(A) as documented in Event Number #49296.

PREVIOUS OCCURRENCES:

A review was performed for similar Pilgrim Station Licensee Event Reports (LERs) submitted to the NRC. The review focused on LERs involving feed pump trip events. The review identified no similar events or LERs.

ENERGY INDUSTRY IDENTIFICATION SYSTEM (EIIS) CODES:

The EIIS codes for Components and Systems referenced in this report are as follows:

SYSTEMS	CODES
None	N/A

REFERENCES:

Condition Report, CR-PNP-2013-5949, Manual Scram due to trip of all three Reactor Feed Pumps.